

Monte Carlo N Particle Extended (MCNPX) Radiation Shield Modelling on Boron Neutron Capture Therapy Facility Using D-D Neutron Generator 2.4 MeV

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ABSTRACT

Based studies were carried out to analyze the internal dose of radiation for workers at Boron Neutron Capture Therapy (BNCT) facility base on Cyclotron 30 MeV with BSA and a room that was actually designed before. This internal dose analyzation included interaction between neutrons and air. The air contained N₂ (72%), O₂ (20%), Ar (0.93%), CO₂, Neon, Krypton, Xenon, Helium and Methane. That internal dose to the worker should be below the dose limit for radiation workers which is an amount of 20 mSv/years. From the particles that are present in the air, only Nitrogen and Argon can change into radioactive element. Nitrogen-14 activated to Carbon-14, Nitrogen-15 activated to Nitrogen-16, and Argon-40 activated to Argon-41. Calculation using tally facility in Monte Carlo N Particle version Extended (MCNPX) program for calculated Neutron flux in the air 3.16x10⁷ Neutron/cm²s. The room design in the cancer facility has a measurement of 200 cm in length, 200 cm in width, and 166.40 cm in height. Neutron flux can be used to calculate the reaction rate which is 80.1x10⁻² reaction/cm³s for carbon-14 and 8.75x10⁻⁵ reaction/cm³s. The internal dose exposed to the radiation worker is 9.08E-9 μSv.

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1. INTRODUCTION

Radiation is the emission of energy in the form of electromagnetic waves or particles. Based on the ability to ionize the material it passes, the radiation is divided into two: ionizing radiation and non ionizing radiation. Non-ionizing radiation is radiation in the form of electromagnetic waves with a wavelength of more than 10 nm and is unable to ionize the material it passes, eg radio waves and visible light. Ionizing radiation is radiation with great energy so that it can ionize the material it passes. This radiation can be electromagnetic waves such as X-rays or particles such as electrons, positrons, alpha protons, neutrons and heavy ions [1,2].

A person can receive radiation doses from natural sources as well as from man-made radiation. In a report published in 2000, UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation), states on average a human will receive a dose of 2.8 mSv (280 mrem) per year [3].

The effects caused by exposure to radiation can vary. In general, the effects are differentiated into two, namely stochastic effects and deterministic effects. Stochastic effects are the effects of radiation that the severity and probability of occurrence have no specific threshold (usually small doses). If the given dose is low and in long time brackets (not all at once) then there is the possibility of cells

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or tissues repairing themselves so that signs of exposure to radiation are not visible. But there is another possibility that the cells are damaged but the former radiation appears after a very long period of time. Examples of stochastic effects are cancer, leukemia, cataracts and genetic disorders. If the radiation dose has exceeded the radiation threshold (high radiation exposure) it will have an increasingly severe damage effect with the increase in radiation dose. Such an effect is called a deterministic effect. An example of a deterministic effect is sunburn at a radiation dose of 2-3 Gy. [2,4,5]

Regulations need to be made so that in the activities of radiation use provide greater benefits from the risks it creates. There are three principles recommended by ICRP in radiation utilization listed in publication 26 and restated in publication 60: [6]

The BNCT study conducted utilizing a neutron D-D Neutron Generator 2.4 MeV source via a beam shaping assembly (BSA). Inside the BSA a collimator is installed to direct the radiation to the room used for in vivo and in vitro tests. This room requires materials that can reduce the radiation that comes out neutrons or gamma so that the radiation dose outside the room is below the applicable threshold limits. The designs made must comply with the applicable regulations, namely Perka BAPETEN number 4 of 2013 on dosage limit values for radiation workers.

2. MATERIALS AND METHODS

2.1 Interaction of Radiation with Matter

Radiation is an emission of energy through matter or space in the form of electromagnetic waves or particles. The decay of radioactive substances emits particles. Particle radiation can be detected by utilizing the radiation interaction with the material [14].

Interactions of Radiation with Matter

Neutron interaction with the material can be divided into two lines, namely scattering or absorption. A more detailed description of these various reactions will be described as follows [16].

Scattering

Scattering is divided into Elastic

Scattering and Inelastic Scattering. Neutrons impinge on an atomic nucleus that is almost always at rest or ground state, i.e. when the atom has the lowest energy under normal circumstances. The neutron comes out of the core and leaves the core in its ground state (not excited). The neutron has undergone elastic scattering by the atomic nucleus because the state of the system remains as it was. The notation of nuclear reactions in this interaction can be written with symbols (n, n).

While inelastic scattering is almost the same as elastic scattering, the difference is that inelastic scattering of nuclei that have been pounded by neutrons is in an excited state, i.e. the atom has more energy than the energy at its ground state. The collision energy is stored in the atomic nucleus so that this interaction is endothermic. Inelastic scatter is symbolized by (n, n').

Absorbed

The catch/absorption reaction is the interaction of nuclei that absorb neutrons and then emit one or more gamma rays called capture γ -rays. This event is an exothermic interaction symbolized by (n, γ). For the symbolized radiation absorption (n, α).

2.2 Interaction of Gamma with Matter

The γ -ray, also known as the photon, is generated by an unstable nuclide source. There are many possible photon interactions with the material, but only 3 important interactions to note in radiation measurements, namely the photoelectric effect, Compton scattering and pair production [1].

Photoelectric effect

The photoelectric effect is the interaction between a photon with an electron that is strongly bonded in an atom that is an electron on the inner shell of an atom, usually K or L skin. Photon will hit the electron and because the electrons are strongly bonded the electrons will absorb the entire photon. As a result, the electrons will be emitted out of the atom with the force of motion as big as the photon's energy difference and the electron binding power.

Compton Scattering

Compton scattering occurs when the X-ray energy unit in the event light is deflected from the original groove by an interaction with an electron. The electron is rejected, removed from its orbital position and the amount of energy depends on the angle or inner angle not spread and on the nature of the diffused medium, since the X-ray energy unit in the diffused light has less energy, and the longer and fewer wavelengths Penetration rather than the unit of energy in light.

Pair production

Pair production is an interaction between a photon and an atomic nucleus. Pair production can only occur if the energy of the coming photon is worth 1.022 MeV or more. The photon energy is completely absorbed, resulting in the photon being transformed into a pair of electron-positron. The rest mass of each electron and positron particle equals to 0.511 MeV. A positron moving in the medium area loses its energy continuously due to collision with atomic electrons. At the end of the track, the positron will recombine with the electrons and both annihilate, then two photons appear with a total energy of $2mc^2$ [1].

2.3 MCNPX (Monte Carlo N-Particle

Extended)

MCNPX is a particle transport simulation technique that is included in theoretical experimentation developed at Los Alamos National Laboratory (US). In the field of sciences, Monte Carlo method is packed in a computer code, one of the most widely used is Monte Carlo N-Particle (MCNP). The Monte Carlo N-Particle (MCNP) computer code is a particle transport code with three-dimensional geometry and source modeling capabilities applicable to reactor physics, criticality shielding, environmental nuclear waste cleanup, medical imaging, and several other related fields. [20,21].

The Monte Carlo method has the ability to perform simulations in various methods: neutrons, photons, neutrons, neutrons, or neutrons-electrons formulated in the input code format. MCNPX is an MCNP code designed to simulate particles with a wide energy range. MCNPX 2.6.0 has a new capability, especially

in transmutation phenomena, burnup and production of kasep particles. Several new tally source options and variance-reductions have also been added. Physical improvements are the new version of the Cascade-Exciton Model (CEM), the addition of the Los Alamos Quark-Gluon String Model (LAQGSM) and a substantial improvement to the muon physics [22].

Tally shows the information that users want to collect. MCNPX software is able to calculate various information such as currents, surface flux, average cell flux, and flux at a particular point in the detector. This tally can be numbered 1, 2, 4, 5, 6, 7, 8 or 10 (e.g., 11, 21, 34, 44 and so on) but should not exceed three digits. Particle designator <pl> is added to the tally which is a symbol of the particle to be studied, e.g., n for neutrons, -n for anti neutrons, p for photons, e for electrons and -e for positrons [26]. In MCNPX randomly determined by what nucleus neutrons will interact, the type and location of the interaction, its direction, the energy and type of particles formed after the interaction.

2.4 Quality Factor of Radiation

Nuclear radiation has different ionization capabilities. High LET radiation has greater damage but shorter range. Whereas low LET radiation has a small damage and a longer range. LET is directly related to the energy transferred in the material. The ability of radiation to cause different effects for human tissue is quantified in the constant of radiation quality factor.

2.5 Soft Tissue

Modeling healthy human tissue on MCNPX simulation using soft tissue. Soft tissue contains elements similar to healthy human tissue. The soft tissue is used to record the dose rate coming out of the radiation shield, which the radiation workers are working out of the room.

This research begins by collecting information and data related to problem which will be discussed that is looking for BSA which will be used, room modeling and material selection through literature study from papers, thesis, book, diktat and from internet.

In solving the problem, this research is divided into several sections, namely;

Use of the MCNPX on Code Making

The use of the MCNPX program in this study was used to determine the rate of absorption of leakage. In order to calculate the dose of neutron and gamma the tally used was F4. The normalizing factor of neutrons and gamma for dose calculation is

$$2,5 \times 10^{11} \quad \text{n/s} \quad (4.3)$$

Calculating the rate of radiation doses of neutrons and gamma requires the conversion of kerma to convert the energy released by neutrons and gamma into dose function.

The conversion calculation from energy to dose is continued by dividing the energy range according to the value of the quality factor to obtain the equivalent dose of the sievert unit.

The soft tissue used in the calculation contains some elements of ICRP (International Commission Radiation Protection) 234 which are converted into MCNPX code.

The room is made simply according to the needs of BATAN (National Nuclear Power Agency) for in vivo and in vitro tests. The main layer is homogeneous to facilitate the calculation of tally. The second layer and the third layer are made as beam catchers.

Identification of problems

This study aims to support the radiation safety system. This study considers the worst possible outcome when any existing safety system cannot function properly.

Plan Result Analysis

The use of simulation of soft tissue with a thickness of 5 cm is the modeling of a person or a pekerja outside the room. The calculation on the soft tissue simulation uses the tally contained in the MCNP program. The tally calculations F14 and F24 will produce the radiation dose values absorbed by the soft tissue, the exit value will be converted into Gy/s units. To get the unit Sv/s then the value obtained was multiplied by the quality factor [15,16].

3. RESULTS AND DISCUSSION

3.1 Beam Shaping Assembly Modelling

Beam shaping assembly (BSA) used in this research is BSA modeling which has been designed by Prayoga with specification as can be seen in Figure 3.1.

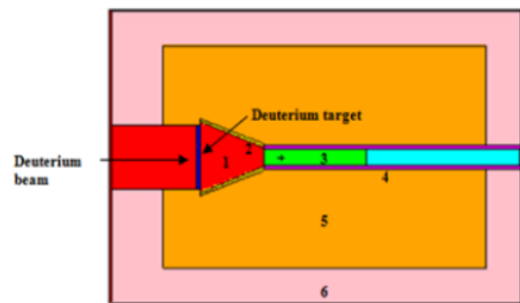


Fig 1. Beam shaping assembly.

The beam shaping assembly as shown above uses a collimator with air material and Nickel (Ni) while the reflector uses elements of Pb and paraffin. For Gamma shield material bismuth (Bi) is used. Then the moderator material used is AlF3.

3.2 Room Modelling

The room is made with a length of 200 cm, a width of 200 cm, and a height of 166.4 cm adapted to the needs of BATAN for in vitro and in vivo tests. The simulated room is just a blank space filled with air. A calculation of doses is made by placing materials such as those contained in soft tissue (healthy tissue) around the room because the radiation is isotropic that spreads in all directions. Figure 3.2. Below clarify the above description:

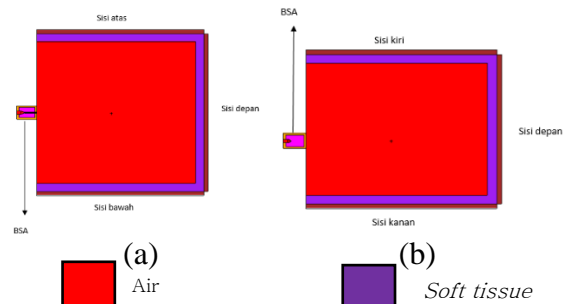


Fig 2. Modeling used to calculate dose rates without shielding: (a) Side View (b) Top View (in cm) [28]

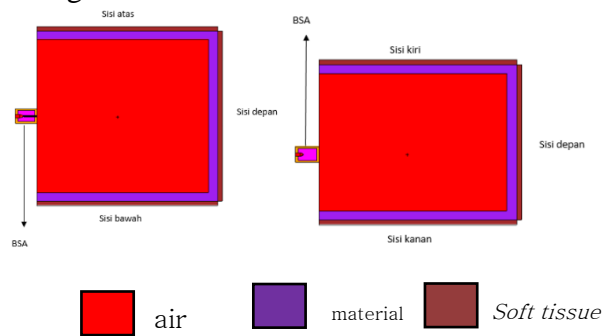
The initial dose obtained after running MCNPX is shown in the Table 1

Table 1. Results of simulated dosage without shielding

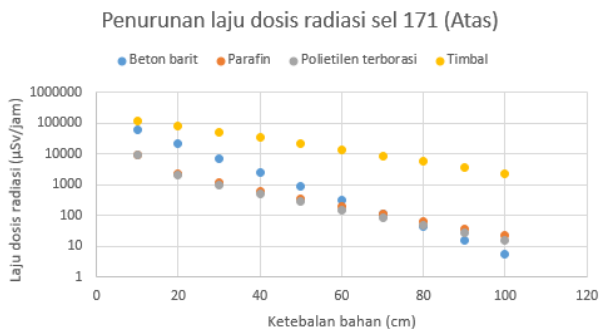
Cell	Dose rate ($\mu\text{Sv}/\text{hour}$)
Above	119811.98
Below	173459.53
Left	121869.23
Right	121655.46
Front	615792.05

3.3 Radiation Shielding Modelling

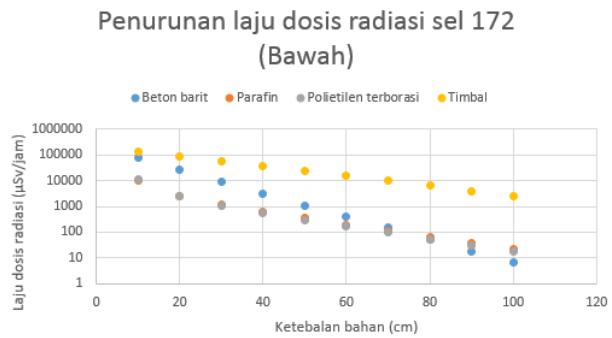
The four materials which had been selected to be simulated are paraffin, barit concrete, lead and polyethylene. The four materials were simulated against the radiation coming out of the aperture with a thickness variation of 10 cm to 100 cm. Modeling of radiation shield can be seen in Figure 3.

**Fig 3.** Simulation of the effect of material thickness on the radiation dose out: (a) Side view (b) Top View

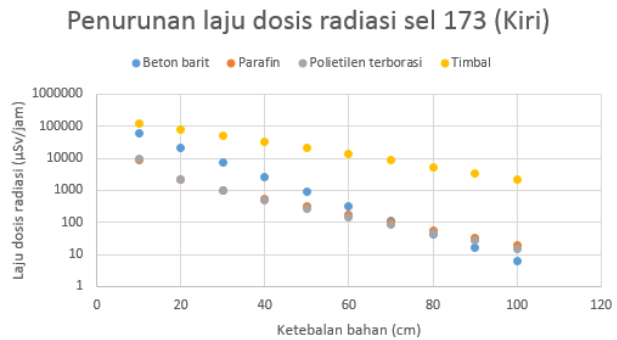
Simulation results using five tally cells are shown in the graph.



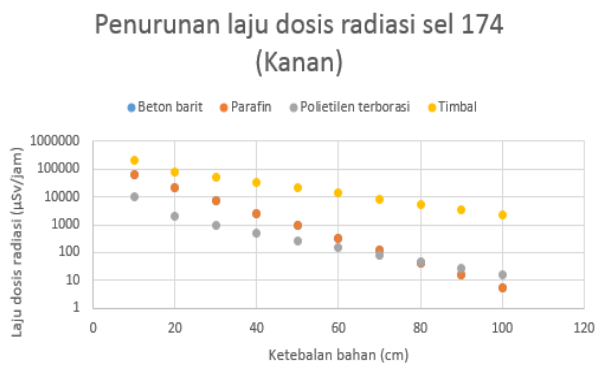
(a) Above cell



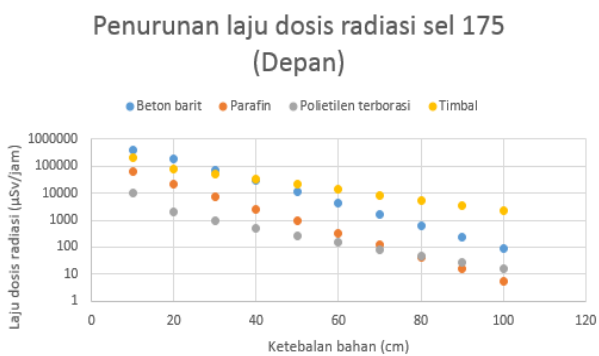
(b) Below cell



(c) Left cell



(d) Right cell



(e) Front cell

Fig 4. Graph of decreasing rate of radiation dose on material thickness

3.4 Radiation Shielding Modelling

1. Design 1

Design 1 (one) uses 90 cm barrels of concrete and 40 cm of polymer as the main layer. Additional layers of 10 cm barrels of concrete and polyethylene are decorated each 10 cm in thickness with a smaller volume.

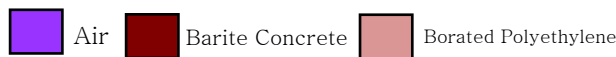
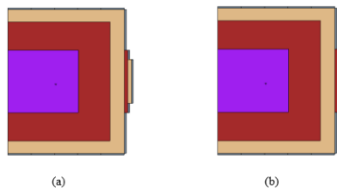


Fig 5. Design of radiation shield with barite concrete and borated polyethylene Material (a) Side view (b) Top view

2. Design 2

Design 2 (two) uses 90 cm barite concrete material and 40 cm paraffin as the main layer. Additional layers of 10 cm barrels of concrete and polyethylene are decorated each 10 cm in thickness with a smaller volume

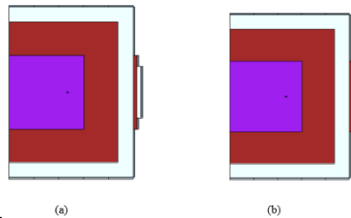


Fig 6. Design of Radiation Shield with Barite and Paraffin Concrete Material (a) Side view (b) Top view

3. Design 3

Design 3 (three) uses 100 cm barrels of concrete and 40 cm of polymerization as the main layer. Additional layers of 10 cm barrels of concrete and polyethylene are decorated each 10 cm in thickness with a smaller volume.

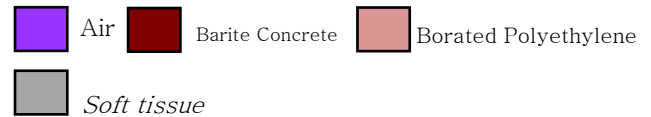
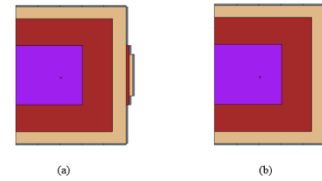


Fig 7. Design of Radiation Shield with Barite Concrete and Borated Polyethylene Material (a) Side view (b) Top view

4. Design 4

Design 4 (four) uses a material of 100 cm barrel concrete and 40 cm paraffin as the main layer. Additional layers of 10 cm barrels of concrete and polyethylene are decorated each 10 cm in thickness with a smaller volume.

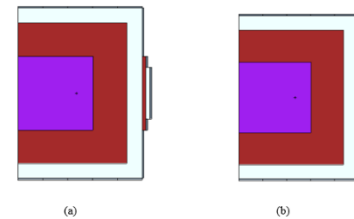


Fig 8. Design of radiation shield with barite concrete and borated polyethylene Material (a) Side view (b) Top view

The four radiation shield designs above meet the upper limit value of the allowed radiation dose of 10 $\mu\text{Sv/hr}$. The most optimal radiation shielding model was chosen from the radiation shield options based on the highest leak rate of radiation dose on each radiation shield design, as well as the average leakage rate of the radiation dose.

Table 2. Summary of The Four Radiation Shielding Designs

	Design 1	Design 2	Design 3	Design 4
The largest dose rate ($\mu\text{Sv}/\text{hour}$)	5.62	7.18	4.58	4.75
Average dose rate ($\mu\text{Sv}/\text{hour}$)	1.408	1.512	0.65	0.683

The simulation calculations of each design indicate that the entire radiation design meets the BAPETEN Perka, the radiation dose not exceeding 10 ($\mu\text{Sv}/\text{h}$). The design that has the smallest radiation dose rate in the cell that has the largest radiation dose rate in each design is the design 3. The smallest average dose rate is in design 3.

4. CONCLUSION AND REMARKS

After doing research of radiation shield modeling of BNCT facility with neutron source accelerator CNG 2.4 MeV the following conclusions can be taken: 1) The best materials to use as a radiation shield at 10-70 cm thickness are parafin and polyetelin terborasi, while at the thickness of 80-100 cm the best material is barit concrete.; 2) The best design of the four designs that have been made is the third design composed of spun polyethylene and barite concrete with a maximum dose rate of 4.58 $\mu\text{Sv}/\text{hour}$ and the average radiation dose rate of 0.65 $\mu\text{Sv}/\text{hour}$.

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